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EXECUTIVE SUMMARY

THE CHARACTERIZATION OF LOW LEVEL WASTE ARISING FROM THE PICKERING LARGE SCALE FUEL CHANNEL REPLACEMENT PROGRAM

J.P. Krasznai

Research Division has over the last three years undertaken a program to characterize low and intermediate level nuclear waste arising from Ontario Hydro's Nuclear Generation Program. Waste characterization is an essential part of Ontario Hydro's waste management strategy. The data generated has potentially many users. It may be used by the designers of new processing equipment (e.g. supercompactor, incinerator), by designers of waste storage facilities and by the Waste Site operators to safely handle, package and store the waste and ensure license compliance for the Waste Operations Site.

This report details the results of a one year program to determine the physical and radiochemical composition of both processable and non-processable waste from the Pickering Large Scale Fuel Channel Replacement Program. In view of the C-14 particulate contamination problem and the cutting and removal of portions of fuel channels, this waste stream was expected to be quite different from normal operating waste.

Over 620 processable waste bags were visually inspected. From the compositional point of view this waste was identical to that from normal operating waste. The radiation field on contact with these bags was usually less than 2 mR/h. The major gamma emitting radionuclides are Co-60 8-20 uCi/bag, Sb-125 1.7-2.5 uCi/bag, Nb-95 1.2-2.0 uCi/bag and Cs-137 1-1.5 uCi/bag. The C-14 in the waste was 315 uCi/bag, which was approximately 2000 times the levels found in incinerable waste (170 nCi/bag), and compactible waste arising from normal operations (100 nCi/bag). The Sr-90 content of each waste bag (1.5 uCi) was comparable to that measured in waste generated as a result of normal reactor operation.

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The presence of transuranic isotopes was difficult to confirm directly from gamma spectra. However, the presence of transuranics in non-processable waste was confirmed by alpha spectrometry and also from swipes taken from Unit 1 and 2 feeder pipes which indicate that there is 6-60 pCi/cm^2 of alpha particulate deposited on the feeders of Units 1 and 2. It is therefore reasonable to assume that processable waste arising from this operation would also be contaminated. Based on correlation factors calculated from alpha spectra of smears of processable waste components that relate Am-241 to the plutonium isotopes, it is estimated that each bag of processable waste contained approximately 60 nCi of alpha particulate.

About 10% of the total low level (<200 mR/h) non-processable waste generated during the retubing program was retrieved from storage and smears taken of the contents. Based on a conservatively determined smearing efficiency of 10%, the specific activity (nCi/cm^2) of all detectable radionuclides present in the waste containers was determined. From an estimation of the total surface area in each container, the radionuclide inventory per m^3 container was estimated.

Alpha activity in this waste ranged from 3-10 uCi/m^3 for yokes/studs, 7-1135 uCi/m^3 for end fitting bodies and 54-1120 uCi/m^3 for end fitting liners. C-14 activity in this waste was also variable; ranging from 4.5-380 mCi/m^3 for yokes/studs, 0.08-1.5 mCi/m^3 for end fittings and 0.08-454 mCi/m^3 for end fitting liners. The major gamma emitting nuclides in the non-processable waste in order of importance are Co-60, Sb-125 and Cs-137.

Approximately 180 m^3 of steel waste associated with reactor face components were generated during the two unit LSFCR program. It is expected that all reactors will be retubed at least once in their 40 year design life which will generate a significant quantity of metallic waste. Reducing the volume of this waste may be economical. Aggressive decontamination techniques are available to remove surface activity. However, the degree of neutron activation in the bulk of the metal is likely to be the limiting factor in deciding whether or not volume reduction of this waste is practical. In light of this it is recommended that the spatial (radial and axial) distribution of the concentration of activation products in these components be carefully determined.

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THE CHARACTERIZATION OF LOW LEVEL WASTE ARISING FROM THE PICKERING LARGE SCALE FUEL CHANNEL REPLACEMENT PROGRAM

J.P. Krasznai

This report deals with the determination of the physical and radiochemical composition of low level processable and non-processable radioactive waste from the Pickering NGS Large Scale Fuel Channel Replacement Program. The highest radiation field associated with this waste was 90mR/h on the boxes containing end fitting liners due primarily to Co-60. The radiation field from other non-processable waste was in the 2-10 mR/h range. Processable waste was less active and usually less than 2 mR/h. Its physical composition was not significantly different from incinerable waste arising from normal reactor operation. Carbon-14 and alpha particulate was found in the retubing waste at levels which are significantly higher than in normal reactor operating waste although well below the level at which the waste has to be considered to be high level waste according to U.S. N.R.C. guidelines. It is recommended that the degree of neutron activation and the distribution of activation products in the metallic waste be investigated to determine whether volume reduction methods are feasible for this waste stream.

1.0 INTRODUCTION

An important aspect of any waste management strategy whether for waste processing, storage or final disposal is the determination

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of the type and quantity of the waste generated and the expected future arisings of these wastes. Research Division has over the last three years undertaken a program to characterize low and intermediate level nuclear waste arising from Ontario Hydro's Nuclear Generation Program. The following waste streams have been characterized with respect to chemical and radiochemical composition: Incinerator ash/1/, Compactable waste/2/ and Incinerable waste/3/. In addition, programs are underway to characterize ion exchange resins, non-processable wastes arising from normal reactor operations and miscellaneous organic liquid wastes.

This report details the characterization of both processable (incinerable and compactable) and non-processable waste arising from the Large Scale Fuel Channel Replacement Program (LSFCRP). As a rationale for this work, we felt that these waste streams would be very different from those arising from normal reactor operations in a number of aspects:

1. Little or no detectable tritium was expected in the waste since both the moderator and heat transport systems were drained of heavy water. In addition, the heat transport system was vacuum dried.

2. Previous work/2/ had assumed that C-14 contamination in waste was associated with heavy water in the carbonate/bicarbonate form as a consequence of the fast neutron activation of the O-17 isotope. However the C-14 contamination from the annulus gas system was in particulate form, not very soluble in either acid or basic solutions and highly dispersible because of the small particle size (<0.45u). While C-14 in incinerable waste is not normally a storage or disposal problem since it is released from the waste as CO_2, it does become a concern if this incinerable waste is compacted because of incinerator problems or an inadequate incinerator capacity.

3. The contribution from activation products was expected to be higher in this waste because of the generation of large quantities of reactor components such as channel restraint devices (yokes, studs nuts and 'C' clamps), end fittings and liner tubes. The cutting of the Zircalloy-2 pressure tubes and end fittings was also expected to generate significant quantities of Sb-125 (from neutron activation of Sn-124, an impurity in the alloy) and Co-60 respectively in processable waste.

4. The cutting and removal of the end fittings and pressure tubes was also expected to release small amounts of tramp uranium associated fission products and transuranic elements (Am-241 and Pu-239/240).

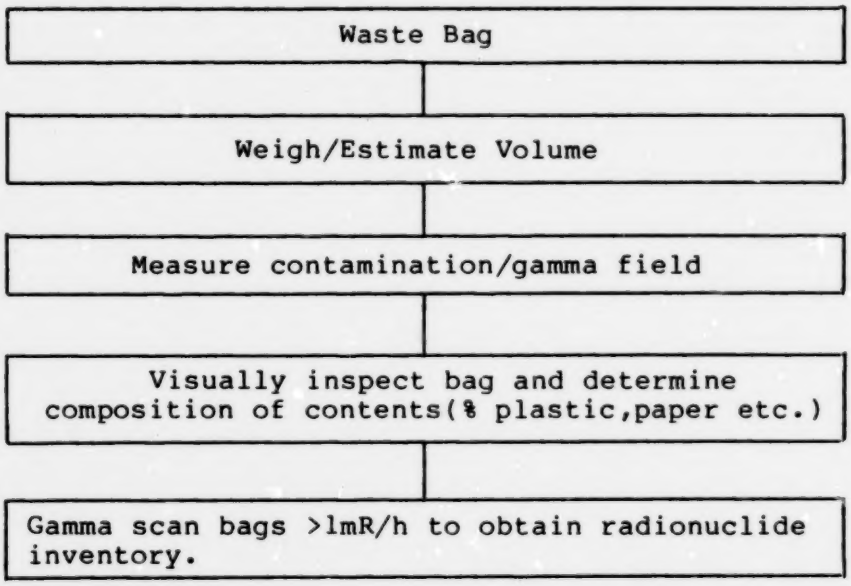
The characterization program was carried out over a period of a year (June '85 to May '86) in order to look at a representative cross-section of the different types of waste.

2.0 EXPERIMENTAL

2.1 Processable Waste

Waste bags which are either incinerable or compactable were identified by Pickering Operations/4/ using blue tape and were double-bagged. At the Bruce Nuclear Power Development (BNPD) Waste Operation Site the bags were unloaded, checked for high gamma fields (>60 mR/h) and non-combustible components and based on these criteria segregated as either incinerable or compactable waste. The protocol followed for the characterization of this waste is given in Figure 1.

FIGURE 1


Waste Bag
Weigh/Estimate Volume
Measure contamination/gamma field
Visually inspect bag and determine composition of contents(% plastic,paper etc.)
Gamma scan bags >1mR/h to obtain radionuclide inventory.

One method used to determine the radionuclide inventory in waste bags was by gamma scanning them, one at a time on a turntable 1 m from the detector. To calibrate the detector for this geometry a simulated waste bag with typical composition of combustible materials was made up containing a mixed source of known activity placed at the center of the bag. This bag was then gamma-scanned at 1m from the detector and the detector efficiencies for each isotope determined.

As mentioned above the determination of the C-14 content in the waste bags became an important parameter following the discovery of extensive C-14 particulate contamination of the reactor vaults originating from the annulus gas system. Work by Analytic Services Section/5/ indicated that the particulate with which the C-14 was associated was only sparingly soluble in either basic or acidic solutions. Consequently the technique used in the

determination of C-14 in compactable waste/2,3/ which relied on the complete leaching of the contamination by NaOH solution would have underestimated the C-14 content of the bags. Complete combustion of the particulate and absorption of the C-14 as CO_2 was the only reliable method for C-14 determination.

The problem of C-14 determination was therefore a problem of obtaining representative samples from a highly heterogeneous waste stream. We looked at a number of methods to homogenize a waste bag prior to taking a sample for C-14 determination by combustion. Freezing the waste in liquid nitrogen followed by manual compaction to obtain particle sizes suitable for small scale combustion was difficult because the waste had to be kept cold during the compaction process which caused a lot of splashi..g. Furthermore the fog created made it difficult to control the operation. Shredding simulated waste bags was attempted/6/ but the process is very aggressive and would require extensive dust control measures which would make this experimental technique prohibitively expensive.

The method that was used for the determination involved segregating approximately 450 incinerable bags from Pickering Units 1&2. These bags were characterized according to the protocol in Figure 1 and subsequently incinerated in two separate burns in the radioactive incinerator at BNPD. A continuous flow of the off$_3$gases (approximately 120 mL/min) from the main stack flow (60 m^3/min) was passed through a heat traced combination particulate/charcoal filter followed by three glass bubblers in series each containing 300 mL 1.5M KOH solution to trap the CO_2 produced during the burn. The effluent from these traps was passed through a CO_2 analyzer to ensure that the traps were effective at scrubbing the off gases of CO_2. Figure 2 shows the general layout of the off gas analysis system which was designed by Analytic Services Section and RMEP Department personnel and operated by them with assistance from the Nuclear Materials Research Unit.

Each bubbler was acidified and the CO_2 liberated was absorbed into Carbasorb solution. A sample of this solution was then withdrawn, mixed with Pico Fluor scintillation cocktail and counted. The method has been described in detail in references 7 and 8.

During off gas sampling the carbon monoxide content in the return gases to the stack was periodically monitored using a Drager tube to ensure C-14 was not moving through the bubblers as incompletely oxidized carbon monoxide which is not trapped. The CO content of the return gas was consistently less than 300 mg/kg/9/.

Some of the bags contained small (nCi) quantities of Am-241. The quantity of other transuranic materials in the waste was determined by analysing the ash from these burns/10/.

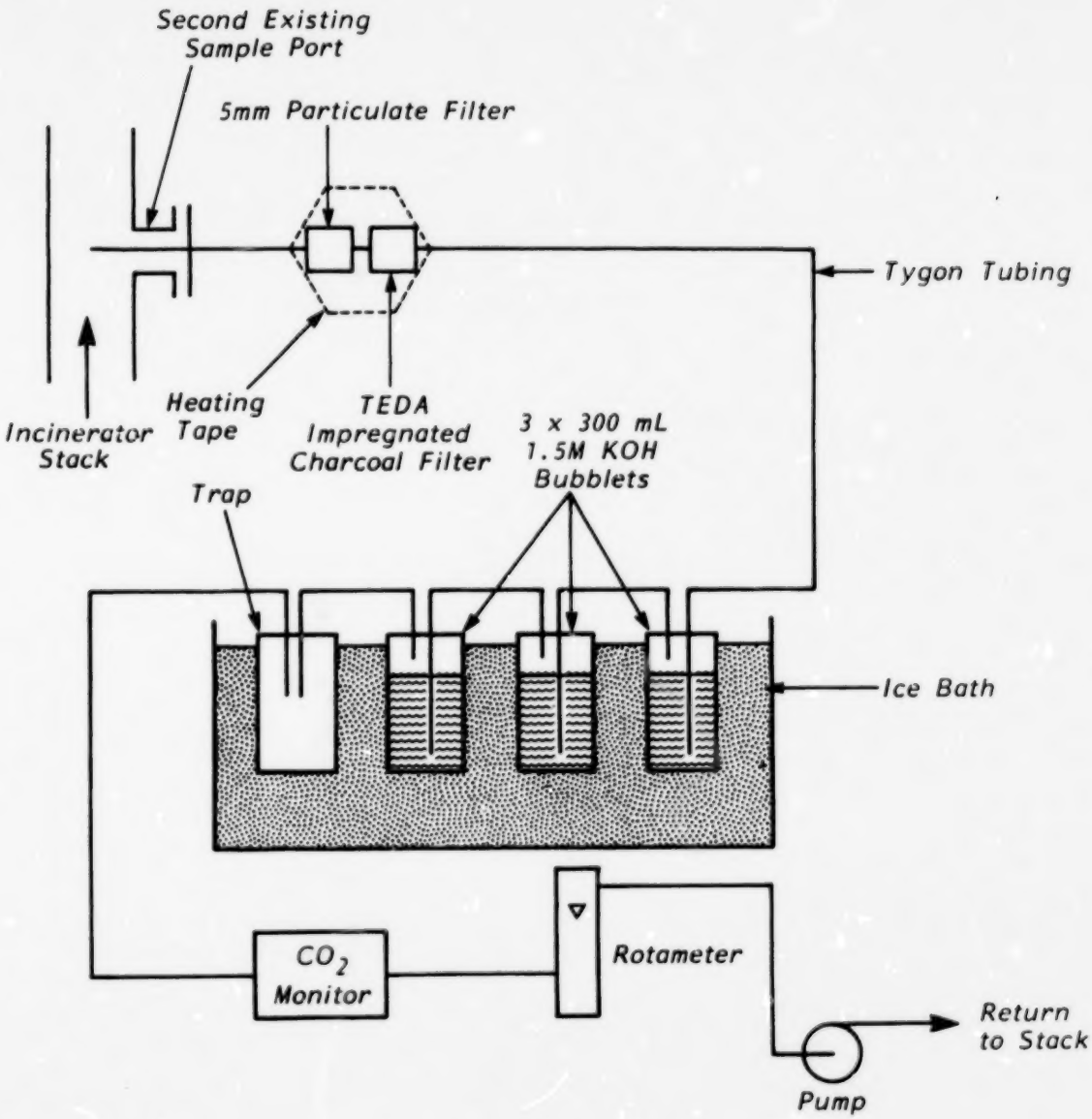


FIGURE 2

INCINERATOR OFF-GAS SAMPLING SYSTEM FOR ^{3}H AND ^{14}C

173037 RD

2.2 Non-Processable Waste

The waste in this category consisted of:

(a) Fuel channel restraint devices such as yokes, studs, nuts, spacers, 'C' clamps and associated sleeves and locking devices.

(b) End fitting sections (outboard of annulus gas bellows).

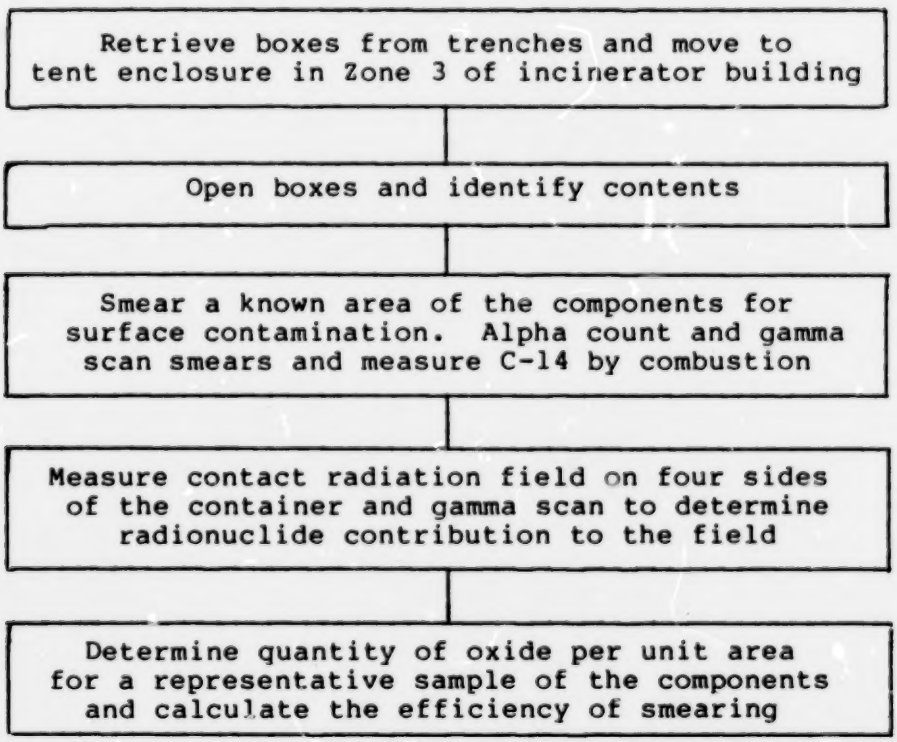
(c) End fitting liner sections (outboard of annulus gas bellows).

(d) Grayloc seal rings and miscellaneous metallic components.

The waste was packaged in 1 m^3 metal containers lined with polyethylene sheeting. With the exception of the boxes containing end fittings and liner tubes the containers were only partially filled. It should be noted that the inboard portions of the end fittings and liner tubes are highly activated and these are stored along with the pressure tubes in the Dry Storage Modules located at Pickering NGS. The waste considered in this work is low level Type I (<200 mR/h).

This waste was characterized using the following protocol:

FIGURE 3


Retrieve boxes from trenches and move to tent enclosure in Zone 3 of incinerator building
Open boxes and identify contents
Smear a known area of the components for surface contamination. Alpha count and gamma scan smears and measure C-14 by combustion
Measure contact radiation field on four sides of the container and gamma scan to determine radionuclide contribution to the field
Determine quantity of oxide per unit area for a representative sample of the components and calculate the efficiency of smearing

The way the measurements are done, the alpha activity data will also include the contribution from Am-241. The procedure for obtaining alpha activity data was to count the smears in a proportional counter. These samples however were invariably greasy and dirty and so whatever the instrument measures is the alpha activity on the surface of the smear. Any activity embedded in the grease or in the paper fibres will be effectively shielded. Consequently this method will underestimate the true alpha activity. To get an idea of the magnitude of this effect, a greasy smear was counted at 1150 Volts on the proportional counter. The same smear was subsequently processed to remove the alpha activity from the sample and deposit it as a thin film on a planchet and recounted. The ratio of the two counts (0.16) shows that only 16% of the activity is counted in the as received matrix. This process however is difficult, time consuming and therefore cannot be used for every smear. Consequently the alpha count rate for each smear was adjusted by this factor to generate the alpha activity data reported in Table 6.

3.0 RESULTS

3.1 Processable Waste

624 bags were examined over a period of about a year and the data has been compiled into a data base. Table 1 shows the average composition of the waste which is mostly paper and plastic. The "average" waste bags also contained about equal amounts of cotton, rubber and other materials such as metal, wood, concrete and floor sweeping compound. The composition of the plastic was mostly polyethylene and polyvinyl chloride (PVC) from disposable plastic suits. The quantity of halogenated rubber such as neoprene was only 1% by volume. As can be seen from the standard deviation column it is somewhat misleading to talk about an "average" composition because of the large variations in composition of the bags. For example, some bags may contain nothing but plastic while some others may contain nothing but paper. The composition of this waste is very similar to that from normal reactor operation where the "average" composition was 39% plastic, 5% rubber, 9% cotton, 40% paper and 7% other materials.

Approximately 440 bags were incinerated in two seperate burns dedicated to Pickering Unit 1 & 2 bags. The off gases were analyzed for tritium and C-14, and samples of bottom and fly ash from these burns were gamma-scanned. Approximately 2 L each of bottom ash and flyash were removed from the top portions of the appropriate ash containers. Since a burn produces 250 L of bottom ash and 15 L flyash, these samples were assumed to be large enough to be considered representative of the ash generated during the burn. 218 of these bags which made up the first burn were included in the 624 bags which were characterized.

Table 1

Composition of Waste Arising From the LSFCR Program

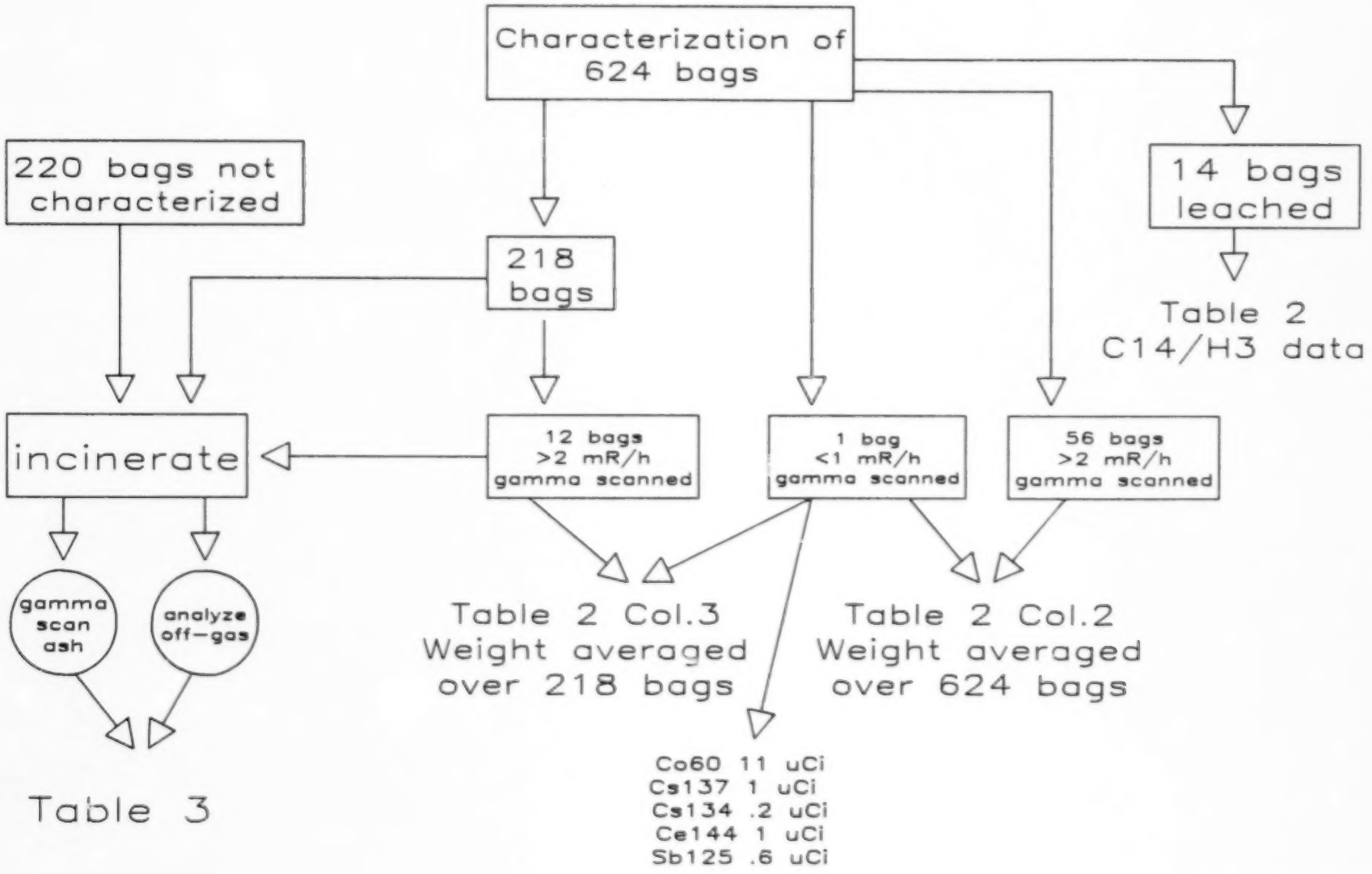
WASTE CONTENTS	% COMPOSITION (BY VOLUME) AVERAGE	MAX.	MIN.	STD. DEV.
PLASTIC	33	100	0	34
PVC 13%				
PE 18%				
PC 2%				
RUBBER	8	100	0	15
NATRUB 6%				
NEOPRENE 1%				
OTHER 1%				
COTTON	5	100	0	13
PAPER	49	100	0	35
OTHER	6	100	0	20

Radionuclide inventory data (Tables 2 and 3) were generated by a number of methods which are summarized in Figure 4 in flowsheet format. However, the statistics and methodology are more reliable for the data presented in Table 3 than for Table 2.

The tritium content in each bag was calculated to be approximately 7 mCi, considerably higher than in a bag of normal incinerable waste/3/ (1mCi). This is somewhat surprising since both the heat transport and moderator systems of Units 1 & 2 were drained prior to the start of the retubing operation and it would be expected that the tritium activity of the waste from these units would reflect this. A possible explanation of this observation is that although the tritium concentration of the Units 1 & 2 moderator systems are more than a factor of 1000 lower than in operating units, (4 mCi/kg and 24 mCi/kg respectively/11/), the proportion of moderator system waste from Units 1 & 2 was greater than the waste from operating units, probably as a result of the Emergency Coolant Injection System backfit. The C-14 in the waste was approximately 2000 times the levels found in incinerable waste (170 nCi/bag)/3/, and compactable waste arising from normal operations (100 nCi/bag) /2/. This is consistent with the C-14 particulate problem that has prevailed during the LSFCR program on both Units 1 & 2. The strontium 90 content of the waste bags was determined from the ash analysis to be 1.5 uCi/bag. This number is not significantly different from that obtained from the analysis of incinerator ash produced from burning normal operating waste/12/.

As expected, the major gamma emitting radionuclides are Co-60 8-20 uCi/bag, Sb-125 1.7-2.5 uCi/bag, Nb-95 1.2-2.0 uCi/bag and

FIGURE 4


Characterization of 624 bags
220 bags not characterized
218 bags
14 bags leached
Table 2 C14/H3 data
incinerate
12 bags >2 mR/h gamma scanned
1 bag <1 mR/h gamma scanned
56 bags >2 mR/h gamma scanned
gamma scan ash
analyze off-gas
Table 2 Col.3 Weight averaged over 218 bags
Table 2 Col.2 Weight averaged over 624 bags
Co60 11 uCi
Cs137 1 uCi
Cs134 .2 uCi
Ce144 1 uCi
Sb125 .6 uCi
Table 3

Cs-137 1-1.5 uCi/bag. The cobalt and cesium activities are considerably lower than in incinerable waste from normal reactor operation (42 and 31 uCi/bag respectively) possibly as a consequence of defueling and the heat transport decontaminations carried out prior to the start of the LSFCR program. Sb-125 ($t_{1/2}$=2.7 years) from the activation of Sn-124, an impurity in Zircalloy 2 pressure tubes and Zircalloy 4 fuel sheaths, is a radionuclide that appeared in all of the waste from Units 1 & 2 probably as a result of the pressure tube cutting operation, and to a lesser extent perhaps from erosion of the pressure tube material during defuelling.

Table 2

PICKERING INCINERABLE WASTE
GAMMA SCAN DATA

RADIONUCLIDE	BAG SCANS uCi/Bag	BAG SCANS uCi/Bag	MEAN uCi/Bag
H-3	467.2	467.2	467.2
C-14	1.77	1.77	1.77
Cr-51	0.04	0.12	0.08
Mn-54	0.03	0.03	0.03
Fe-59	0.24	0.13	0.19
Co-60	12.60	12.29	12.45
Zn-65	ND	ND	ND
Zr-95	0.15	0.13	0.14
Nb-95	0.20	0.21	0.20
Ru-103	0.18	0.06	0.12
Ru-106	0.22	0.15	0.19
Sn-113	ND	0.01	0.01
Sb-124	0.26	0.09	0.18
Sb-125	0.89	1.46	1.18
Cs-134	0.15	0.15	0.15
Cs-137	0.89	1.09	0.99
Ce-141	0.03	0.01	0.02
Ce-144	0.86	0.95	0.91
Am-241	0.01	0.03	0.02
Pu-238	0.005	0.01	0.011
Pu-239+Pu-240	0.017	0.05	0.034

ND : Not Detectable

The presence of transuranic isotopes is difficult to confirm directly from gamma spectra, because Am-241, the only gamma emitting transuranic nuclide has a single gamma peak at 59.5 kEV which is difficult to distinguish from the X-ray peaks of the other isotopes. However, the presence of transuranics in non-processable waste was confirmed by alpha spectrometry and also from swipes taken from Unit 1 and 2 feeder pipes/13/ which indicate that there is 6-60 pCi/cm^2 of alpha particulate deposited on the feeders of Units 1 and 2. It is therefore reasonable to assume that processable waste arising from this operation would also be contaminated. From the results of the

alpha spectrum the ratio of Am-241 to Pu-238 was 1.9 and Am-241 to Pu-239+Pu-240 was 0.59. The ratio of Am-241 to total alpha activity was 0.3. These ratios were used to generate the plutonium data in Table 2 from the measured Am-241 activity.

Table 3

PICKERING 1&2 INCINERABLE WASTE
DEDICATED BURN DATA

RADIONUCLIDE	BURN 86-79 uCi/Bag[a]	BURN 86-80 uCi/Bag	MEAN uCi/Bag
H-3	9174.31	4587.16	6880.73
C-14	350.92	280.73	315.83
Cr-51	0.24	0.28	0.26
Mn-54	0.15	0.59	0.37
Fe-59	0.06	0.06	0.06
Co-60	8.13	27.78	17.96
Zn-65	0.07	0.02	0.05
Sr-90	1.49	not done	1.49[b]
Zr-95	0.65	1.09	0.87
Nb-95	1.23	1.98	1.61
Ru-103	0.16	0.18	0.17
Ru-106	0.72	0.66	0.69
Sn-113	0.01	0.03	0.02
Sb-124	0.01	0.00	0.01
Sb-125	1.74	2.45	2.10
Cs-134	0.15	0.22	0.19
Cs-137	1.05	1.51	1.28
Ce-141	0.07	0.14	0.11
Ce-144	0.66	0.56	0.61

(a) Calculated from the ash analyses (Report No CE86-2577-K). To convert from nCi/g ash to uCi/bag of incinerable waste, multiply by 0.64 (Appendix 2).

(b) S. Sood (Analytic Services) C87-12-K Feb. 1987.

3.2 Non Processable Waste

A total of 18 1 m^3 boxes were characterized. The % Total column in Table 4 refers to the sample population of each type of component examined. For example it is estimated based on the packing efficiency of the boxes examined that there are 70 boxes containing yokes and studs in waste storage of which 9 or 13% were examined.

Table 4

SUMMARY OF NON-PROCESSABLE WASTE CONTAINERS ($1m^3$)

NO OF BOXES	CONTENTS	% TOTAL	AVERAGE CONTACT FIELD (mR/h)
9	Yokes and Studs	13	2
6	End Fitting Bodies	14	10
1	Plugs, Rods and Misc. Components.	?	2.5
2	End Fitting Liners	9	90

As Table 4 indicates, the end fittings and the liners are the most active of the components removed. The boxes were gamma-scanned on all sides at a distance of 1m from each side. However weak gamma emitters would be attenuated to the extent that they would not be detectable by this method and so smears were obtained of the box contents. The smears were gamma-scanned, alpha-counted and then combusted to obtain the radionuclide inventory per unit area smeared. Tables 5 and 6 summarize the results of the external gamma scans and the scans of the smears respectively. The data takes into account shielding by the box contents by dividing the unshielded source term by a suitable shielding factor which was estimated for each radionuclide detected based on an average box density of 600 kg/m^3 and the mass attenuation coefficient of steel at various gamma ray energies. The observation of the weak gamma emitting isotopes Ce-144 and Eu-154 in the boxes containing liners and end fittings indicates that these isotopes are probably not shielded to the extent assumed by the shielding factors. Therefore the activity levels of weak gamma emitters are conservatively estimated.

Some of the more manageable pieces of waste were brought back to Research for analysis. Table 7 summarizes the data for yoke nuts which are representative of yokes and studs. From the point of view of deposited activity this data is assumed to also represent the external surface of end fittings. Table 8 summarizes the data for end fitting liners which are assumed to represent the internal surface of end fittings also. By gamma scanning the components followed by gamma scanning a smear of a fixed area of the component surface, the efficiency of the smearing process was determined. The efficiency was found to be dependent on both the radionuclide and also the component itself which is a reflection of the nature of the surface oxide. The smear efficiency ranged from 10-20% for Co-60, Mn-54 and Sb-125 and approximately 40% for Ce-144, Cs-134, Cs-137 and Ru-106. For the sake of conservatism, the data presented in table 6 used a uniform efficiency for all nuclides of 10%.

Table 5

ACTIVITY (mCi) OF BOX CONTENTS BASED ON EXTERNAL GAMMA SCANS

Isotope	Yokes/studs			E/F Bodies			E/F liners	Others
	Avg	High	Low	A	H	L	A	A
Mn-54	-	-	-	0.8	1.5	0.2	-	-
Co-60	1.4	1.8	1.1	10.0	11.9	8.1	29.8	1.3
Zn-65	-	-	-	0.8	1.6	0	-	-
Zr-95	0.01	0.04	0	-	-	-	-	-
Nb-95	0.02	0.07	0	-	-	-	-	-
Ru-106	-	-	-	1.3	1.9	0.8	-	-
Sb-124	-	-	-	-	-	-	-	0.5
Sb-125	0.8	2.5	0	1.6	1.9	1.4	-	-
Cs-134	-	-	-	-	-	-	-	0.4
Cs-137	0.2	0.3	0.04	0.3	0.3	0.3	-	6.2
La-140	0.01	0.02	0	-	-	-	-	-
Ce-144	-	-	-	0.03	0.05	0.01	-	-
Eu-154	-	-	-	0.001	0.003	0	-	0.002
TOTAL	2.4	4.7	1.2	14.9	19.2	10.8	29.8	8.4

Note: Empty cells in the table denote that the isotope is not observed.

The end fitting data in Table 6 demonstrates the unreliable nature of the alpha activity. Clearly the alpha activity cannot be less than the Am-241 activity. The effect of self shielding by the matrix was taken into consideration when calculating the alpha activity from the alpha count rate by dividing the latter by a correction factor of 0.16. It would appear that the correction factor which was assumed to be a constant in fact varies from sample to sample. Examining individual data entries where both Am-241 and alpha activity is measurable, indicates that Am-241 contributes to approximately 20-30% of the total alpha activity. Similarly there is an approximate 1:1 correlation between the Am-241 and Eu-154 levels. These factors provide a means of estimating the alpha activity from either the Am-241 or Eu-154 activities without having to use the less reliable alpha activity measurement.

One of the boxes contained miscellaneous components that appeared to be plugs of some kind with a diameter that suggested that they were inserted into the fuel channel in the early stages of the program. One such component had the highest alpha activity (1.3 nCi/cm^2) of all the components examined with the exception of one of the end fitting liner samples. However, only one such plug was present in the box and it was not representative of the contents.

Table 6

SPECIFIC ACTIVITY(nCi/cm*2)
BASED ON ANALYSIS OF SMEARS TAKEN FROM COMPONENTS (*).

ISOTOPE	YOKES AND STUDS			END FITTINGS			E/F LINERS			OTHER
	AVERAGE	HIGH	LOW	AVERAGE	HIGH	LOW	AVERAGE	HIGH	LOW	
Alpha	0.003	0.01	0.00	0.03	0.04	0.02	1.3	2.3	0.17	1.3
C-14	371.3	1890.4	0.00	1.57	4.35	0.25	935.7	1420.0	451.4	1.0
Mn-54	0.003	0.03	0.00	0.05	0.12	0.00	0.2	0.3	0.0	-
Co-60	0.795	1.27	0.32	12.17	18.83	3.69	20.4	40.4	0.4	0.8
Zn-65	0.010	0.09	0.00	0.09	0.20	0.00	0.0	0.1	0.0	-
Ru-106	0.033	0.30	0.00	3.67	8.89	0.00	1.3	2.6	0.0	1.2
Sb-125	0.719	4.50	0.02	1.29	2.26	0.38	1.8	3.6	0.0	2.2
Cs-134	0.002	0.01	0.00	0.07	0.17	0.00	0.1	0.2	0.0	3.6
Cs-137	0.169	0.80	0.02	3.82	9.64	0.42	3.6	7.2	0.0	276.0
La-140	0.000	0.00	0.00	0.01	0.04	0.00	0.0	0.0	0.0	-
Ce-144	0.069	0.40	0.00	0.83	1.23	0.26	2.1	4.1	0.0	0.6
Eu-154	0.000	0.00	0.00	0.36	0.93	0.00	0.2	0.4	0.0	3.8
Am-241	0.002	0.01	0.00	0.27	0.67	0.02	0.3	0.7	0.0	3.8

(*) A 10% smearing efficiency has been applied to generate this data.

Table 7

SPECIFIC ACTIVITY OF YOKE NUTS

Isotope	Whole Nut nCi	nCi/cm^2
C-14	-	23[a]-72[b]
Mn-54	40.0	0.17
Co-60	2250	9.40
Zn-65	20.0	0.08
Sb-125	1610	6.70
Cs-134	10.0	0.04
Cs-137	60.0	0.25
Ce-144	40.0	0.17

(a) Calculated from the average oxide thickness 0.003 g/cm^2.
(b) Average contamination on P1 and P2 feeder pipes (Report No CE-86-5540-K).

Table 8

SPECIFIC ACTIVITY OF E/F LINER SECTIONS

Isotope	nCi/cm^2		
	Average	High	Low
C-14	0.2	0.25	0.15
Mn-54	1.8	2.7	0.9
Co-60	297	497	110
Ru-106	4.9	8.9	1.1
Sb-125	8.5	12.3	3.9
Cs-134	-	-	-
Cs-137	0.6	1.0	0.2
Ce-144	2.3	4.0	0.9

Note: Empty cells denote that the isotope is not observed.

Based on the contents of each 1 m^3 container (Appendix 1) it is estimated that the surface area including the internal surface of the container is 20 m^2 for the boxes containing yokes and studs, 32 m^2 for end fitting liners and 34 m^2 for the boxes containing end fitting bodies. Using these figures with the data from Tables 5,6,7 and 8 gives a range of values for the radionuclide inventory in each container. Only the maximum value measured is listed in Table 9.

Table 9

MAXIMUM ACTIVITY (uCi) MEASURED IN $1m^3$ CONTAINERS

Isotope	Yokes/Studs	End Fittings	Liners
Alpha(*)	10	1135	1120
C-14	$3.8x10^5$	$1.5x10^3$	$4.5x10^5$
Mn-54	34	$1.5x10^3$	860
Co-60	$1.9x10^3$	$12.0x10^3$	$1.6x10^5$
Zn-65	18	$1.6x10^3$	32
Zr-95	40	n.d.	n.d.
Nb-95	70	n.d.	n.d.
Sb-125	$2.5x10^3$	$1.9x10^3$	$3.9x10^3$
Cs-134	1	58	66
Cs-137	300	$3.3x10^3$	$2.3x10^3$
La-140	20	14	n.d.
Ce-144	80	418	$1.3x10^3$
Eu-154	n.d.	315	130
Am-241	2	227	224

n.d. - not detectable
* - Estimated using 5:1 ratio of alpha:Am-241

The data is consistent with the radiation field measurements which showed that the field increased in the order end fitting liners > end fittings > yokes/studs. It should be noted that while the processable waste characterized was representative of waste generated over a year of the maintenance effort, the non processable waste components were removed from the reactor face prior to the shock heating process used to free the pressure tubes from the end fittings. It was during this process that the majority of the C-14 contamination was mobilized.

Alpha and C-14 particulate present the most serious radioactive hazard by virtue of their long (>200 year) half lives. Based on an average density of 600 kg/m^3 per container, the maximum alpha activity (estimated from the measured Am-241 level) and C-14 particulate inventory is 17 pCi/g and 0.6 uCi/g respectively in those containers with yokes and studs. In the boxes containing end fitting bodies the inventory was 2 nCi/g and 2.5 nCi/g for alpha and C-14 respectively. The most gamma active container with end fitting liners had an alpha and C-14 inventory of 2 nCi/g and 0.75 uCi/g respectively.

The U.S Nuclear Regulatory Commission has developed regulations governing the disposal of radioactive waste in shallow land burial sites (10CFR.61). Part of these regulations provide a classification scheme for waste ie Class A, B or C. There are concentration limits for the radionuclides which determine which class the waste falls into. For example, the Class A limit for C-14 is 0.8 Ci/m^3 and for alpha emitting radionuclides with a half life greater than 5 years it is 10 nCi/g. This class of waste can be disposed of with a minimum of packaging. If the C-14 and alpha content of the waste is greater than 10 times these limits, i.e 8 Ci/m^3 and 100 nCi/g respectively, then the waste is unsuitable for near surface disposal. These regulations are very similar to the proposed standards of the U.S. Environmental Protection Agency/14/ which provide concentration limits for specific radionuclides above which a waste is defined as high level waste. For alpha emitting radionuclides with a half-life greater than 20 years this limit is 100 nCi/g. The proposed limit for C-14 is 8 uCi/g. The Pickering LSFCRP non-processable waste is below these proposed limits but within a factor of two of being classified as Class C waste under U.S regulations. The implication of this, assuming the level of uncertainty in the data is at least a factor of two, is that the void volume inside the containers will need to be filled before the waste can be sent for disposal.

The estimated volume of waste steel generated per reactor is approximately 90 m^3. It is very probable that each reactor will be retubed at least once during its 40 year design life. In view of this there may be an economic incentive to consider technologies which would permit release of this waste as non radioactive such as that being developed by RMEP and Research Division/15/ for stainless steel fuel storage baskets, where the surface activity must be less or equal to 0.1 uCi/m^2 after decontamination. There is an additional problem associated with the waste stream from the Retubing operation, in that the base metal being continuously exposed to a neutron flux is therefore slightly activated. Although the degree of activation is expected to be low (5 nCi Co-60/g)/16/ compared with the activity associated with the metal oxide (>300 nCi Co-60/g), a number of key questions will need to be answered before volume reduction techniques can be applied to this particular waste stream. For example how much activation of steel takes place and how are the activation products distributed both axially and radially in the steel components? The most important question of course is at what level of activity can the components be considered to be non radioactive? Some work has been done to determine activation levels in both Bruce/17/ and Pickering end fitting liners/16/. In both cases Cr-51 was the most abundant nuclide followed by Co-60. The data indicated a sixfold decrease of activity over a distance of 1m from the end shield. This suggests that even if chemical decontamination of the steel is impractical, separating the higher activity portion of the components by cutting may be feasible.

4.0 SUMMARY

(1) The composition of processable waste from the LSFCR program is very similar to that found from normal reactor maintenance operations. The waste is composed mostly of paper (49%) and plastic (33%) with an equal distribution of rubber, cotton and other miscellaneous items.

(2) The radiochemical composition of the processable waste was determined by a number of different methods. The method considered to be the most reliable involved the dedicated burn of only P1/P2 waste, and the analysis of the off gases followed by an extensive radiochemical analysis of both the bottom and baghouse ash.

(3) Each bag contains 7 mCi H-3 and 0.3 mCi C-14 which is approximately 2000 times greater than measured for the C-14 content of normal operating waste. The Sr-90 content of the waste is 1.5 uCi/bag which is within a range found in waste generated during normal reactor operations. The major gamma emitting nuclides in each bag are Co-60 (18 uCi), Zr/Nb-95 (2.5 uCi), Sb-125 (2.1 uCi), Cs-137 (1.3 uCi), Ru-106 (0.7 uCi) and Ce-144 (0.6 uCi). The total alpha activity per bag based on correlation factors between Am-241 and Pu-238 and Am-241 and Pu-239+Pu-240 is 65 nCi.

(4) If these bags cannot be incinerated for any reason the bags must be compacted to obtain some volume reduction. There are approximately 120 bags per m^3 of compacted waste(600-1200kg). Consequently the radionuclide inventory per m^3 of this waste assuming it is compacted prior to storage is:

H-3	0.8 Ci
C-14	36 mCi
Total Alpha	7.8 uCi
Co-60	2.2 mCi
Zr/Nb-95	0.3 mCi
Sb-125	0.2 mCi
Cs-137	0.2 mCi
Ru-106	0.1 mCi
Ce-144	0.1 mCi

(5) 18 1 m^3 containers containing various reactor face components were retrieved for the purposes of characterization from the trenches at BNPD. This represents approximately 10% of the total amount of Type I (<200 mR/h) non-processable waste generated and is considered to be statistically adequate.

(6) Radiation fields on contact with the containers were 2 mR/h, 10 mR/h and 90 mR/h for boxes containing yokes/studs, end fitting bodies and end fitting liners respectively.

(7) The radionuclide content of the waste was determined by collecting smears, direct gamma scanning of the waste containers and where practical, removing small representative samples from the containers for analysis of activity.

(8) Alpha activity in this waste ranged from 3-10 uCi/m^3 for yokes/studs, 7-1135 uCi/m^3 for end fitting bodies and 54-1120 uCi/m^3 for end fitting liners.

(9) C-14 activity in this waste was also variable; ranging from 4.5-380 mCi/m^3 for yokes/studs, 0.08-1.5 mCi/m^3 for end fittings and 0.08-454 mCi/m^3 for end fitting liners.

(10) The major gamma emitting nuclides in the non-processable w ste in order of importance are Co-60, Sb-125 and Cs-137.

(11) The Pickering LSFCRP waste is well below the limits proposed by both the U.S. N.R.C and Environmental Protection Agency defining waste acceptable for near surface disposal with respect to alpha and C-14 content. However, the non-processable waste with respect to its C-14 content is within a factor of two of being classified as Class C waste under U.S. regulations which would require that the void volume in the waste containers be filled.

5.0 RECOMMENDATIONS

This type of large scale rehabilitation program generates a large volume of active waste. The radiochemical characteristics of the waste are dependent on the materials of contruction and to a large extent on the operating history of the reactor generating the waste. In view of this, it is recommended that this characterization program be repeated for each reactor undergoing such a maintenance effort.

In light of the large volume of metallic waste that is generated during a large rehabilitation program such as the LSFCR project, decontaminating the waste such that it can be treated as inactive may be economically attractive. It is recommended that the degree of activation as well as activation product distribution in the steel components be determined to assess the practicality of this method of volume reduction.

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APPENDIX 1

NON-PROCESSABLE WASTE CONTAINER CONTENTS

Box No	Contents	Radiation Field (mR/h)
P-29	20 YOKES.40 STUDS, ASSOCIATED NUTS AND RINGS.	2.0
P-27	15 YOKES,30 STUDS, ASSOCIATED NUTS AND RINGS.	2.5
P-26	30 YOKES, 60 STUDS, 30 RINGS, APPROX. 250 NUTS.	1.5
P-23	9 YOKES, 18 STUDS, GRAYLOC SEALS, 30 10" OD SS RINGS.	1.5
P-22	25 YOKES, 50 STUDS, 25 RINGS AND ASSOCIATED NUTS.	1.5
P-25	18 YOKES, 36 STUDS AND ASSOCIATED NUTS AND RINGS.	1.5
P-21	25 YOKES, 50 STUDS, ASSOCIATED NUTS AND RINGS.	3.0
P-71	18 YOKES,36 STUDS AND ASSOCIATED NUTS AND RINGS.	2.5
P-81	30 YOKES,60 STUDS AND ASSOCIATED NUTS AND RINGS.	2.0
P-24	15 END FITTING BODIES (WITHOUT LINER TUBES).	7.0
# 21	18 END FITTING BODIES (WITHOUT LINER TUBES).	12.0
P-11	18 END FITTING BODIES.	12.0
# 7	18 END FITTING BODIES.	9.0
#16	20 END FITTING BODIES.	11.0
P-1	50 LINER TUBES.	97.0
P-7	21 LINER TUBES.	85.0
#99	12 STEEL RODS (3'x6") THREADED AT ONE END. 50% OF BOX CONTAINED 12" OD PIPE, 3 FEET LONG CUT LENGTH-WISE. 4" DIAM AND 12" LONG STEEL PLUGS WITH THREADED BLIND HOLES AT CENTER. BOX ALSO CONTAINED PIPE HANGERS AND A STEEL PLUG 8" DIAM THREADED WITH A PIPE HANGER WELDED TO IT.	2.5

APPENDIX 2

CALCULATION TO CONVERT ASH ANALYSIS DATA TO ACTIVITY CONTAINED IN INCINERABLE WASTE BAGS

Assumptions

(1) 15 m^3 incinerable waste produces 0.25 m^3 bottom ash. (L. Krochmalnek, Private Communication, July 18, 1986.)

(2) Density of ash = 730 kg/m^3 (A.S. Williamson, OHRD Report No 83-207-K, May, 1983.)

(3) 15 m^3 waste is equivalent to 250 waste bags.

(4) 15 m^3 waste produces 15 L of baghouse ash.

(5) In the first dedicated Pickering burn (86-79) 218 bags were incinerated.

Therefore, to convert nCi/g of activity in bottom ash to equivalent activity in the waste bag (nCi/bag):

nCi/g(ash) x 1000 g/kg x 730 kg/m^3 x 0.25 $m^3/250$ x 218/250 = nCi/bag

636 nCi/g = nCi/bag
0.64 nCi/g = uCi/bag

Similarly, to convert baghouse ash activity to activity in the waste bags:

nCi/g(ash) x 1000 g/kg x 730 kg/m^3 x 0.015/250 x 218/250 = nCi/bag

38.2 nCi/g = nCi/bag
0.038 nCi/g = uCi/bag

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